THREE MILE ISLAND ACCIDENT

12/4/2015

1. INTRODUCTION

The Three Mile Island (TMI) Accident at Harrisburg, Pennsylvania in the USA is a severe and expensive incident that has seriously affected, and will continue to affect the peaceful development of Nuclear Energy worldwide.

It occurred in the early hours of the morning at 4:00 a.m. on March 28, 1979, at Unit 2 (Fig.1) of the Metropolitan Edison utility's plant, while it was operating at 97 percent of full power. The plant consisted of two Pressurized Water Reactors (PWRs) each of an electric capacity of 961 MWe.

The accident took 14 years to clean up at a cost of $1 billion.

Figure 1. Metropolitan Edison's unit 2 Pressurized Water Reactor (PWR) of 961 MWe of electrical capacity is the capped cylindrical building between its two cooling towers in the foreground. Unit 1 in the back was unaffected by the accident and continued operation.

In the history of technology, severe accidents in some cases meant a total disavowment of the technology. For instance, the sinking of the Titanic meant the end of the era of large ocean liners. The Hindenburg Zeppelin accident meant the end of air transportation using Hydrogen filled dirigibles. Unlike these, nuclear electrical generation has continued to be used in the USA after the TMI accident, with marked improvements in the safety of existing nuclear power plants.

However, in spite of the fact that many nuclear plants have been constructed in other parts of the world, none has been built since then in the USA. It is clear that any construction of new nuclear power plants in the USA will depend on more innovative designs than the existing ones, incorporating passive and inherently safe safety features.

The accident is here described with an attempt is made at identifying the lessons that could be learned from it.

2. DESCRIPTION OF PLANT

The reactor is a Pressurized Water Reactor (PWR) operating at a pressure of about 2,000 pounds per square inch (psi). The primary coolant is ordinary water, H_2O flowing from the reactor to the steam generator. Typically the outlet temperature is at 325 degrees C, and the inlet temperature is at 292 degrees C.

The steam generator is of the once-through flow design type, designed and manufactured by the Babcock and Wilcox Company. The coolant enters from the top of the steam generator, flows through vertical tubes in it , and exits at its bottom. Reactor coolant pumps pump water from the bottom of the steam generator, back to the reactor pressure vessel. This primary system is a closed loop, with the reactor coolant contained in two loops, designated as loop A and loop B, during normal operation.

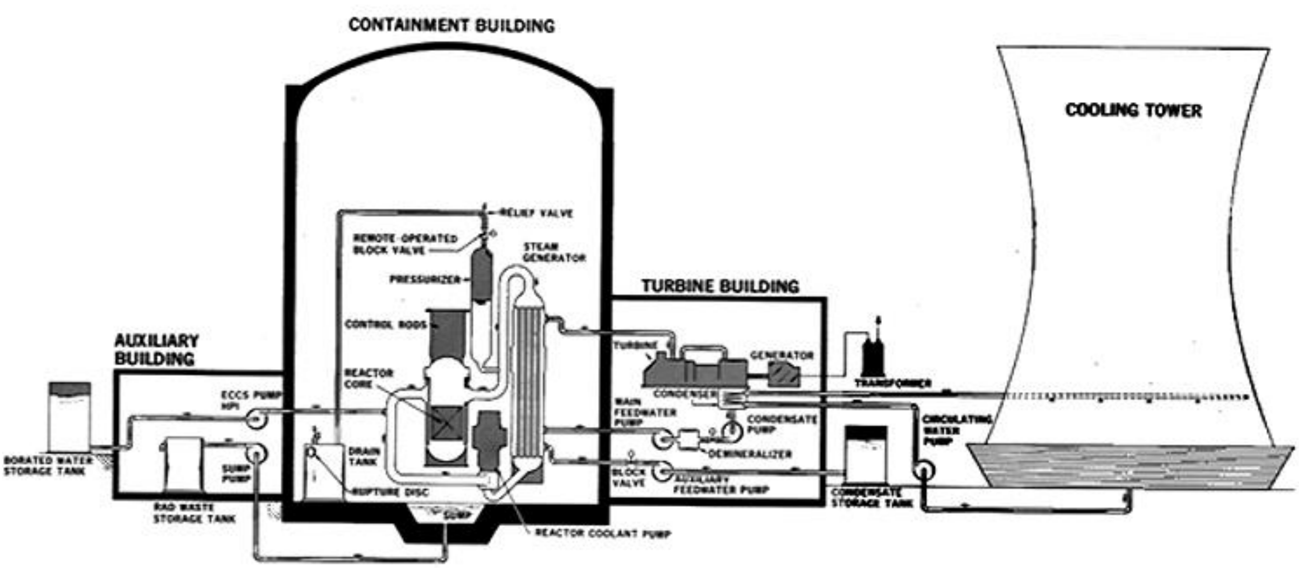

Figure 2. Outline of the Three Mile Island Unit 2 (TMI-2) power plant.

To prevent water from boiling in the primary loop, it is pressurized using a pressurizer unit. It is partially filled with water, and contains electrical heaters at its bottom. Upon turning on the heaters, steam is generated within the pressurizer, and exerts a pressure on the surface of the water in the pressurizer. This pressure is transmitted to the rest of the primary system. Spray valves on top of the pressurizer can spray cold water into the steam region, quenching it and subsequently reducing the pressure in the primary system. To avoid over-pressurization of the primary loop, a bank of relief valves with predetermined set points are positioned on top of the pressurizer. Some of these are Power Operated Relief Valves (PORVs), controlled by the reactor operator. If actuated, steam is discharged from the pressurizer to a drain tank situated at the bottom of the reactor building. This drain tank has a rupture disk that releases the contents of the tank if it overflows to the containment sump. A remotely operated block valve is available, should the PORV valve fail to reseat itself after releasing the pressure.

In the secondary system, feed water is pumped and flows outside the steam generator's tubes. Heat is transferred from the water on the primary side to water on the secondary side, where it is allowed to turn into steam. This steam flows to the steam turbines which rotate and produce electricity in an electrical generator. The steam that leaves the turbine is condensed into liquid in the condenser unit. After passing through a demineralizer, it is pumped back into the steam generator by the feed-water pumps.

As a safety feature, in case the main feed water system becomes unusable, auxiliary feedwater pumps are available to supply cooling water from the condensate storage tank to the steam generators. A block valve controls the flow of the auxiliary feed water to the steam generators.

Another safety feature is the reactor's Emergency Core Cooling System (ECCS). A High Pressure Injection System (HPIS) operates automatically if the primary system's pressure decreases below 1,600 psi. It pumps water from a storage tank into the cold leg of the primary system. If the operator wishes, the high pressure injection system can pump water containing boric acid into the core. Boron would absorb any neutrons and shut down the chain reaction in case the control rods cannot be inserted into the core to shut down the chain reaction.

3. ENGINEERED SAFETY FEATURES (ESFs)

In general, the ECCS for PWRs has the following installed safety systems:

1. **The accumulators.** These are large vessels containing water under nitrogen pressure. They are connected to the primary system through automatic valves which open and inject water into the primary system should the pressure fall below about 40 bars.
2. **The High Pressure Injection System (HPIS)**. This allows pumping of water at pressures of about 100 bars, but at a relatively low rate.
3. **The Low Pressure Injection System (LPIS)**. This allows water to be pumped at a high flow rate, at a pressure below 30 bars.
4. **Containment Spray.** It is operated by the LPIS pumps and gets water from the Refuelling Water Storage Tank (RWST).
5. **Decay Heat Removal System.** Provides a long term cooling means for the reactor core, through the Component Cooling Water System (CCWS). It uses the LPIS pumps, and circulates any water accumulating in the containment sump.
6. **Control rods.** These are inserted automatically into the core to shut down the chain reaction, in case certain pressure, core water level, or temperature limits are violated.

7. **Boron Injection.** Borated water is injected into the core in case the control rods cannot be inserted into the core.

4. SEQUENCE OF EVENTS

It is typical of accident occurrences, nuclear or non nuclear, that accidents are initiated by an initiating event, followed by a sequence of other events which could be caused by:

1. Human errors
2. Design flaws
3. Equipment failures.

The initiating event by itself may not be serious enough in most situations, since it allows recovery by means of the design features incorporated into the engineering system by its designers. Only when the other, possibly unanticipated, components of the sequence occur, does a simple upset evolve into a serious accident situation or a "beyond-design-basis" accident.

4.1 INITIATING EVENT

(4:00 A.M., MARCH 28, 1979)

Plant personnel were performing a routine maintenance on the feed water system. A condensate pump moving water from the condenser to the demineralizer was inadvertently stopped. Due to the ensuing loss of suction pressure, the main feed water pumps automatically stopped, as it is designed to do.

This is a benign initiating event from which recovery would have been simple according to the plant design. In this case, the auxiliary-pump would have operated bringing in water to the steam generator. This did not occur, because the block valves downstream of the feed-water pumps had been left closed in a previous maintenance activity.

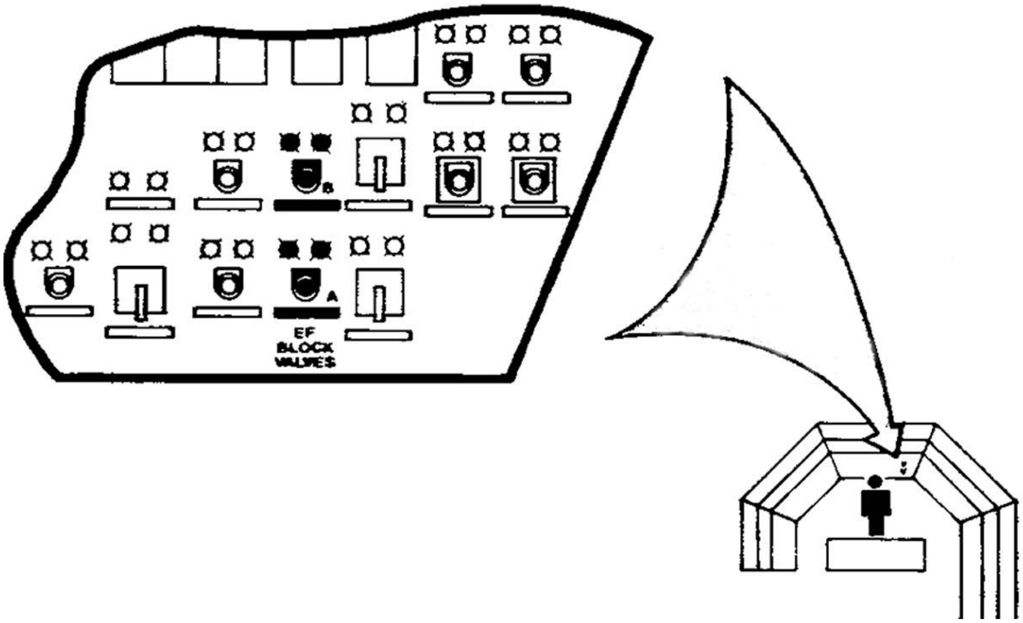

Figure 3. Position of the auxiliary feedwater system block valves on the main operation console.

The operators did not notice that these valves were left closed because of the layout of the main control console. These valves must always be left open during normal operation. Their open position is indicated by a red light above the valve position switch as shown in Fig. 3. The operators did not recognize that they were left closed until 8 minutes into the accident. Tags added at some distant past maintenance event obscured them, and the previous shifts operators did not warn the subsequent operators about their presence.

4.2 TURBINE AND REACTOR TRIP

(0-6 MINUTES)

As a result of the stoppage of the main feed water pumps, the turbine-generator shut down. The auxiliary feedwater pumps started automatically, as designed to act. These pumps would have provided the needed emergency feed-water to the steam generators. This did not happen since the block valves were left closed through a human error.

More heat was being generated in the primary system than could be extracted by the steam generators which were starved of water. As a result the water coolant in the core and primary system increased in temperature, and expanded in volume into the pressurizer. This compressed the steam on top and increased the primary system's pressure. The pressure increased so much in the pressurizer that the safety valves on top of the pressurizer opened to relieve the pressure, as designed at about 4 seconds into the event.

Pressure continued to increase in the reactor vessel until about 8 seconds, at which time, the reactor's automatic control system sensed the pressure and temperature increases, and automatically inserted the control rods, shutting down the fission chain reaction, and most of the power generation.

Following the reactor's scram, the primary system's pressure started decreasing. The pressurizer water level also decreased, as expected in such a type of a transient. At this point, the pressurizer relief valve should have reseated itself and should have closed. However, an equipment failure occurred, in which case this valve did not reseat itself and remained open. As it remained open, it continued to discharge steam into the drain tank at the bottom of the containment building.

A human error now occurred. The operators failed to notice until 2 1/4 hours into the accident that this valve was still open. Should have they realized this; they could have closed a remotely operated block valve that is provided for such a situation.

A design flaw was such that the closure of the valve was indicated by the energization of the solenoid that activates it, not by the actual stem position of the valve. Since the solenoid was energized, the operators thought that the valve was closed. This design flaw was later corrected with the sensors indicating the actual stem position in the valve rather the energization of the actvating solenoids.

Within 2 minutes the primary system pressure continued to drop to 1,600 psi. The two HPIS pumps in the ECCS started automatically as designed, providing sufficient cooling to the core. The pressurizer level started to increase at this point, to the point of almost filling it up.

The operators were trained not to allow the water ingress into the pressurizer to turn it “solid” (the use of the word "liquid" would be more appropriate), leading to a loss of control of the primary system's pressure, since there would be no condensible steam there to reduce the pressure. Unaware that the relief valve on top of the pressurizer was still open, the operators committed a human error in shutting down the HPIS pumps, thinking that this would avoid completely filling the pressurizer with water. One pump was shut down at 4 1/2 minutes, and the second one was also shut down at 10 1/2 minutes.

4.3 LOSS OF COOLANT

(6-20 MINUTES)

The inappropriate emphasis on the pressurizer level indication alone, which led to the premature shutting-down of the HPIS pumps, is considered the most crucial human error in the accident. Up to this point, recovery to a safe shut down, was still possible. If the HPIS pumps were allowed to perform their designed safety function, in keeping the core covered with water in the event of a small-break Loss Of Coolant Accident (LOCA) such as the stuck open relief valve.

The HPIS pumps being shut off, and the relief valves being stuck open, pressure continued to drop in the primary system and water in the warmest parts of the core flashed into steam and boiled off. This situation is analogous to when the cap on the radiator in a car is unknowingly opened while the radiator is still pressurized. The coolant in the radiator, being normally under pressure remains in the liquid phase at the saturation temperature corresponding to the operational pressure. Once it senses atmospheric pressure by the opening of the cap, hot water flashes into steam at below its saturation pressure, with a subsequent loss of the coolant from the core.

Even though the main fission reaction was shut down by the insertion of the control rods, heat continued to be generated from the decay heat, afterheat or afterglow process. This decay heat initially amounts to about 6-7 percent of the operational fission power, and decreases rapidly after shutdown, but needs to be extracted nevertheless by the decay heat removal system in the plant. The negative beta and gamma decays of the fission products produced in the fuel by the fission process generate it. Should the heat not be extracted, the coolant would continue to be lost in the form of steam.

4.4 INITIAL DEPRESSURIZATION

(20 MINUTES -2 HOURS)

The reactor's four main coolant pumps were still running, but started to show high shaft vibration signs from the pumping of a water-steam mixture. The main coolant pumps are designed to pump primarily a single phase liquid coolant. Fearing damage to the main reactor coolant pumps, the reactor operators shut off 2 reactor coolant pumps at 73 minutes and the remaining 2 pumps at 100 minutes. These actions terminated any forced cooling of the steam water mixture in the reactor primary system.

4.5 CORE HEAT UP

(2-6 HOURS)

In the core, without any forced circulation cooling, water continued to be turned into steam and be lost from the system as a result of the decay heat generation, without its being replenished in any way. From 2 to 4 hours into the accident, most of the water boiled away. The temperature of the fuel cladding increased from the normal 625 degrees F to about 2,000-3,000 degrees F. This heat up led to fuel cladding oxidation causing core damage.

At 2 hours 18 minutes into the accident, the operators realized that the block valve associated with the stuck open PORV valve could be closed. Still at this point, the system could have been repressurized and cooling reestablished, but the opportunity was lost.

Once the block valve was closed, the reactor's pressure increased. A site area emergency was declared at 2 hours 55 minutes into the accident, after high radiation levels were detected in the line connecting the reactor coolant circuit to the purification system. It became clear that the core became uncovered by the coolant, the cladding was damaged, and the fission products contained in fuel matrix and the cladding were being released into the primary system.

One main cooling pump in loop B was activated for 19 minutes but later shut down due to excessive vibration and cavitation. The peak fuel temperature at 3,000 degrees C was reached at 3 hours into the accident. The HPIS was reactivated at 3 hours 20 minutes, quenching the fuel, and recovering the core. This terminated the core heat-up phase of the accident.

A general area emergency was declared at 3 hours 30 minutes, once radioactivity started increasing in the reactor, the auxiliary, and the fuel handling buildings. The containment building had high level of activity.

4.6 CONTINUED DEPRESSURIZATION

(6-11 HOURS)

To provide a high coolant flow rates, the operators tried to reduce the pressure in the primary system so as to activate the accumulators and the LPIS. This was started at 7 hours 58 minutes, by opening the PORV block valve. At 8 hours 41 minutes, the pressure reduced to 41 bars, initiating the accumulators. The water injected was insufficient to fully cool the core.

Some of the zirconium fuel cladding oxidized in the high temperature steam environment according to the reaction:

$$Zr + H_2O \rightarrow ZrO + H_2 \qquad (1)$$

This reaction leads to the release of hydrogen gas. The hydrogen gas accumulated in the upper parts of the reactor and formed a hydrogen bubble. This bubble became a cause of concern, since hydrogen is explosive in the presence of oxygen through the water forming reaction:

$$2H_2 + O_2 \rightarrow 2H_2O. \qquad (2)$$

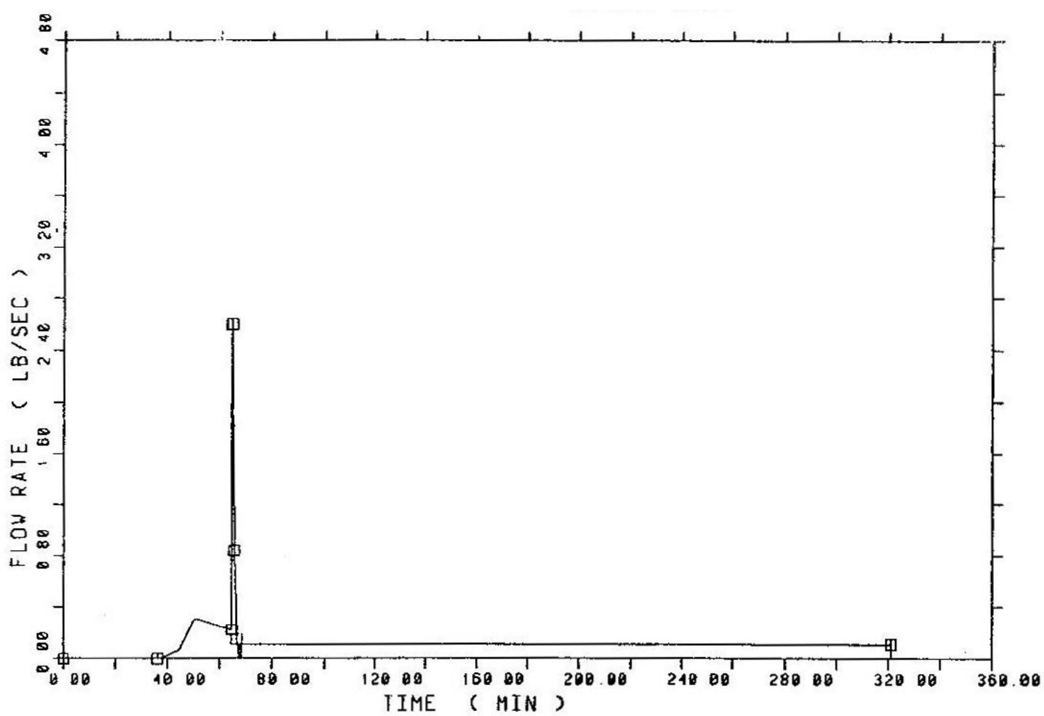

Figure 4. Hydrogen release in the Three Mile Island accident. Pressure pulses were detected in the containment structure from possible hydrogen ignition. The containment structure withstood the pressure pulses. The inclusion of hydrogen recombiners in containment structures was adopted in the USA as a result of this event.

At 9 hours 50 minutes, a pressure pulse probably due to a hydrogen explosion in the containment building, leading to the activation of the containment spray pumps. These were shut down within 6 minutes.

The lowest pressure achieved by the effort of the operators was about 30 bars, and it remained above the pressure of 28 bars at which the high flow rate LPIS could be activated. The block valve was reclosed at 11 hours 8 minutes, after the attempts at depressurizing the system failed. For 2 hours, there was no effective way of removing the decay heat from the reactor core. Injection of coolant using the HPIS was at a low flow rate. Water was being lost through the line to the water purification system.

4.7 REPRESSURIZATION, STABLE COOLING

(13-16 HOURS)

Thirteen hours and 30 minutes into the accident, the PORV valve was reclosed. Sustained HPIS operation was possible, which repressurized the system. This allowed the circuit pumps to pump water from the heat exchanger to the core. At 15 hours and 51 minutes into the accident, a main circulating pump was restarted, leading to a stable heat rejection mode.

4.8 HYDROGEN BUBBLE

(1-8 DAYS)

About a metric tonne of hydrogen was produced by the zirconium-steam reaction. It accumulated at the top part of the reactor vessel, and posed an explosion hazard if it were allowed to mix with oxygen. The gas was being absorbed in the coolant at about 70 bars. This water was bled into a tank kept at atmospheric pressure, from which the gas evolved. It was passed through a system delaying its release for 30 days, to allow other radioactivity to decay, and then vented through the off-gas stack to the atmosphere. In addition, the heaters in the pressurizer were turned on, forcing the dissolved gas in the coolant into the pressurizer's gas space. The block valve on top of the pressurizer was opened to allow its escape.

Natural circulation was achieved at about one month after the accident on April 30, 1979, and the main pumps were turned off. At that point their frictional heating was larger than the decay heat emitted by the fission products.

5. CONSEQUENCES OF ACCIDENT

At the peak temperature that was reached at 2,000 degrees F, most of the cladding oxidized. As a result, the pellets slumped and formed a rubble bed. This rubble bed is hard to cool since it offers 200-400 times the flow resistance of the unaffected core.

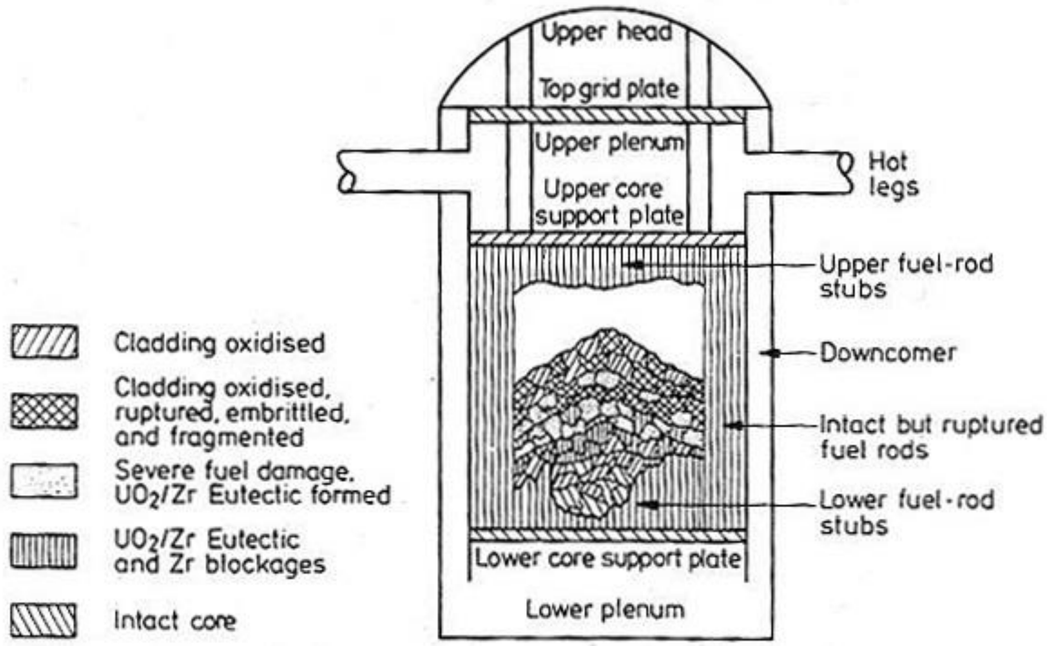

Figure 5. Schematic of debris bed formed in the Three Mile Island Accident.

Krypton^{85} gas with a half life of 10 years was released, in addition to xenon. About 10,000 curies of Krypton^{85} had to be vented from the containment a year after the accident. A minute amount of 16 curies of the iodine released from the fuel was released from the containment. No significant exposure of the public to radioactivity occurred. It is calculated that there will be less than one additional cancer death due to the accident in a total of 200,000 normal cancer deaths in the surrounding population within the subsequent 30 years.

The population dose to 2 million people living within 50 miles of the plant has been estimated as 3,500 [persons.cSv] or [persons.rems]. This amounts to about 1.75 mrems per person of dose equivalent. This radiation exposure is compared to other sources of exposure in Table 1.

Technically, the following factors were considered as significant in the course of the accident:

1. The auxiliary feedwater valves being left closed; a human error.
2. The pressurizer relief valve failing to close; an equipment failure.
3. The false indication of the closure of the valve; a design flaw.
4. The failure to recognize the failure to close of the relief valve for 2 and 1/4 hours, a human error.
5. The inappropriate emphasis on the pressurizer water level indication, a human error.
6. The premature closure of the HPIS cooling, a human error.
7. The shutdown of the reactor coolant pumps, a human error.
8. The failure of containment isolation, where the contaminated water in the reactor building was pumped out to the auxiliary building, a design flaw.

An investigative commission suggested the following factors as contributing to the accident:

1. Inadequate operators training, as noticed from the preponderance of human errors into the accident's sequence of events.
2. Inadequate control room design.
3. The complacent attitude toward safety in the USA nuclear power industry at the time of the accident.
4. Inadequate regulation by the USA nuclear Regulatory Commission (NRC).

The dose received by members of the public is about 1 percent of the average received by any member of the USA's population in a given year.

The end result is a loss of the plant, along with costly recovery operation over a number of years, and an opportunity loss for the power that the plant could have produced over its lifetime.

The concept of defense in depth of a power plant using multiple barriers against the release of radiation was successful in protecting members of the public and the plant's operators. However, erroneous human intervention prevented the engineered safety features from carrying out their intended design functions.

This allowed a situation of a manageable transient of the anticipated small loss of coolant accident type, to escalate beyond the limiting fault condition or the design basis accident.

As a lesson from the accident numerous design modification were implemented in existing nuclear power plants worldwide. Better personnel training and a culture of safety was instigated in the nuclear power industry. However, the effects of human errors have precluded the construction of new power plants. The consensus is that any future plant designs must involve passive safety features that minimize the intervention of the operators with the safety features, based either on lack of knowledge and training, or on complacency and overconfidence,

Table 1. Comparison of radiation exposures from the TMI-2 accident

Source	Dose Equivalent [cSv/(person.year)] [rem/(person.year)]
From Natural Background Radiation	
Harrisburg, Pennsylvania	0.100
Denver, Colorado	0.150
USA average	0.124
From medical radiation sources	
Chest X-ray	0.030
Gastro Intestinal X-ray	0.210
USA average	0.055
From the TMI-2 accident	
Maximum	0.080-0.085
Average	0.001-0.002

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2. John G. Collier and Geoffrey F. Hewitt, "Introduction to Nuclear Power," Hemisphere Publishing Corporation, Springer-Verlag, 1987.